I. ENDF/B REACTION TYPES

The following partial list includes some of the more useful reactions for use with the FMn input card and with the cross-section plotter (see pages 3–93 and B–14.) The complete ENDF/B list can be found in the ENDF/B manual.[1] The MT column lists the ENDF/B reaction number. The FM column lists special MCNP reaction numbers that can be used with the FM card and cross-section plotter.

The nomenclature between MCNP and ENDF/B is inconsistent in that MCNP often refers to the number of the reaction type as R whereas ENDF/B uses MT, but they are the same. The problem arises because MCNP has an MT input card used for the $S(\alpha,\beta)$ thermal treatment. However, the nomenclature between Monte Carlo transport and Deterministic transport techniques can be radically different. See Reference 2 on page G–74 for more information.

Generally only a subset of reactions is available for a particular nuclide. Some reaction data are eliminated by MCNP in cross-section processing if they are not required by the problem. Examples are photon production in a MODE N problem or certain reaction cross sections not requested on an FM card. FM numbers should be used when available rather than MT numbers. If an MT number is requested, the equivalent FM number will be displayed on the legend of cross-section plots.

Neutron Continuous-energy and Discrete Reactions:

MT	FM	Microscopic Cross-Section Description
1	−1	Total (see Note 1)
2	−3	Elastic (see Note 1)
16		(n,2n)
17		(n,3n)
18		Total fission (n,fx) if and only if MT=18 is used to specify fission in the original evaluation.
	−6	Total fission cross section. (equal to MT=18 if MT=18 exists; otherwise equal to the sum of MTs 19, 20, 21, and 38.)
19		(n,f)
20		(n,n'f)
21		(n,2nf)
22		(n,n'α)
28		(n,n'p)
32		(n,n'd)
33		(n,n't)
38		(n,3nf)
51		(n,n') to 1st excited state
52		(n,n') to 2nd excited state
.		.
90		(n,n') to 40th excited state
91		(n,n') to continuum
101	−2	Absorption: sum of MT=102-117 (neutron disappearance; does not include fission)
102		(n,γ)
103		(n,p)
104		(n,d)
105		(n,t)
106		(n,^{3}He)
107		(n,α)

In addition, the following special reactions are available for many nuclides:

202	−5	total photon production
203		total proton production (see Note 2)
204		total deuterium production (see Note 2)
205		total tritium production (see Note 2)
206		total ^{3}He production (see Note 2)
207		total alpha production (see Note 2)
301	−4	average heating numbers (MeV/collision)
	−7	nubar (prompt or total)
	−8	fission Q (in print table 98, but not plots)

S(α,β):

MT	FM	Microscopic Cross-Section Description
1		Total cross section
2		Elastic scattering cross-section
4		Inelastic scattering cross-section

Neutron and Photon Multigroup:

MT	FM	Microscopic Cross-Section Description
1	−1	Total cross section
18	−2	Fission cross section
	−3	Nubar data
	−4	Fission chi data
101	−5	Absorption cross section
	−6	Stopping powers
	−7	Momentum transfers
n		Edit reaction n
202		Photon production
301		Heating number
318		Fission Q
401		Heating number times total cross section

Photoatomic Data:

MT	FM	Microscopic Cross-Section Description
501	−5	Total
504	−1	Incoherent (Compton + Form Factor)
502	−2	Coherent (Thomson + Form Factor)
522	−3	Photoelectric with fluorescence
516	−4	Pair production
301	−6	Heating number

Electrons (see Note 3):

MT	FM	Microscopic Cross-Section Description
	1	de/dx electron collision stopping power
	2	de/dx electron radiative stopping power
	3	de/dx total electron stopping power
	4	electron range
	5	electron radiation yield
	6	relativistic β^2
	7	stopping power density correction
	8	ratio of rad/col stopping powers
	9	drange
	10	dyield

11	rng array values
12	qav array values
13	ear array values

Notes:

1. At the time they are loaded, the total and elastic cross sections from the data library are thermally adjusted by MCNP to the temperature of the problem, if that temperature is different from the temperature at which the cross-section set was processed (see page 2-29). If different cells have different temperatures, the cross sections first are adjusted to zero degrees and adjusted again to the appropriate cell temperatures during transport. The cross-section plot will *never* display the *transport* adjustment. Therefore, for plotting, reactions 1 and -1 are equivalent and reactions 2 and -3 are equivalent. But for the FM card, reactions -1 and -3 will use the zero degree data and reactions 1 and 2 will use the transport-adjusted data. For example, if a library evaluated at 300° is used in a problem with cells at 400° and 500°, the cross-section plotter and MT $= -1$ and MT $= -3$ options on the FM card will use 0° data. The MT $= 1$ and MT $= 2$ options on the FM card will use 400° and 500° data.

2. The user looking for total production of p, d, t, ^{3}He and ^{4}He should be warned that in some evaluations, such processes are represented using reactions with MT (or R) numbers other than the standard ones given in the above list. This is of particular importance with the so-called "pseudolevel" representation of certain reactions which take place in light isotopes. For example, the ENDF/B-V evaluation of carbon includes cross sections for the (n,n'3α) reaction in MT $= 52$ to 58. The user interested in particle production from light isotopes should check for the existence of pseudolevels and thus possible deviations from the above standard reaction list.

3. Two electron transport libraries, el and el03, are maintained. The electron transport algorithms and data in MCNP where adapted from the ITS code.[3] The el library was developed and released in 1990 in conjunction with the addition of electron transport into MCNP4; the electron transport algorithms and data correspond (roughly) to that found in ITS version 1. The el03 library[4] was developed and released in 2000 in conjunction with upgrades to the electron physics package; these upgrades correspond (roughly) to that of ITS version 3. The MT numbers for use in plotting the cross-section values for these tables should be taken from the Print Table 85 column headings and are not from ENDF.